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PG&E Letter DCL-03-152

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, DC 20555-0001

Docket No. 50-275, OL-DPR-80
Docket No. 50-323, OL-DPR-82
Diablo Canyon Units 1 and 2
Response to NRC Request for Additional Information Regarding License Amendment Request 01-08, "Credit for Automatic Actuation of Pressurizer Power Operated Relief Valves; Pressurizer Safety Valve Loop Seal Temperature"

Dear Commissioners and Staff:

Pacific Gas and Electric (PG&E) Letter DCL-02-115, dated September 24, 2002, submitted License Amendment Request (LAR) 01-08, "Credit for Automatic Actuation of Pressurizer Power Operated Relief Valves; Pressurizer Safety Valve Loop Seal Temperature." LAR 01-08 would modify Technical Specification (TS) 3.4.11, "Pressurizer Power Operated Relief Valves (PORVs)," and the licensing basis to credit automatic actuation of the Class 1 PORVs, instead of the pressurizer safety valves (PSVs), to limit reactor coolant system (RCS) pressure changes for the spurious operation of the safety injection system at power event, and other design basis accidents. Also, TS 3.4.10, "Pressurizer Safety Valves," would be revised to allow PSV loop seal temperatures to be less than the lower design temperature during plant heatup and cooldown in Mode 3, and in Mode 4 when any RCS cold leg temperature is greater than the low temperature overpressure protection arming temperature specified in the pressure temperature limits report, provided at least one Class I PORV is available and capable of providing automatic pressure relief. This would allow gradual stabilization of the loop seal temperatures, and avoid having to partially drain the loop seals to establish the proper PSV inlet temperature.

On May 28, June 5, and October 1, 2003, the NRC staff requested additional information required to complete their review of LAR 01-08. PG&E's responses to the staff's questions are provided in Enclosure 1.

This additional information does not affect the results of the technical evaluation and no significant hazards consideration determination previously transmitted in PG&E Letter DCL-02-115.

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If you have any questions or require additional information, please contact Stan Ketelsen at (805) 545-4720.

Sincerely,

D. H. Oatley

David H. Oatley
Vice President and General Manager - Diablo Canyon

jer/3664
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UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

	)	Docket No. 50-275
In the Matter of	)	Facility Operating License
PACIFIC GAS AND ELECTRIC COMPANY	)	No. DPR-80
	)	
Diablo Canyon Power Plant	)	Docket No. 50-323
Units 1 and 2	)	Facility Operating License
	)	No. DPR-82

AFFIDAVIT

David H. Oatley, of lawful age, first being duly sworn upon oath states that he is Vice President and General Manager - Diablo Canyon of Pacific Gas and Electric Company; that he has executed this response to the NRC request for additional information on License Amendment Request 01-08 on behalf of said company with full power and authority to do so; that he is familiar with the content thereof; and that the facts stated therein are true and correct to the best of his knowledge, information, and belief.

David H. Oatley
Vice President and General Manager - Diablo Canyon

Subscribed and sworn to before me this 21st day of November, 2003.

Notary Public
County of San Luis Obispo
State of California

CHUCK MACKEY
Commission # 1397547
Notary Public - California
San Luis Obispo County
My Comm. Expires Feb 1, 2007

PG&E Response to NRC Requests for Additional Information Regarding License Amendment Request 01-08, "Credit for Automatic Actuation of Pressurizer Power Operated Relief Valves; Pressurizer Safety Valve Loop Seal Temperature"

The NRC staff provided Questions 1 through 11 on May 28, 2003, and Question 12 on June 5, 2003. Pacific Gas and Electric (PG&E) provided the staff with draft answers on September 18, 2003. The NRC staff provided clarifications or additional questions for Questions 3, 4, 6, 7, 8, and 9 on October 1, 2003.

NRC Question 1

The licensee proposed change to TS 3.4.10, "Pressurizer Safety Valves", would require that at least one Class I PORV available for automatic pressure relief while the PSVs are considered as inoperable due to low PSV loop seal temperature. Since the PORVs are active components which are subject to single active failure following an event, please explain how this proposed change will satisfy the single failure criteria for system required to mitigate a design basis transient or accident while the plant is operating in Modes 3 & 4 and the RCS cold leg temperature is greater than LTOP arming temperature.

PG&E Response

Mode 3

The two Class 1 power-operated relief valves (PORV) are only credited for the mitigation of design basis events in Modes 1, 2 and 3 when the reactor coolant system (RCS) cold leg temperature is greater the low temperature overpressure protection (LTOP) arming temperature. Since Technical Specification (TS) 3.4.11 requires that all three PORVs be operable in Modes 1, 2, and 3, this ensures that the two Class 1 PORVs are available for mitigation and meet the single failure criteria for these modes, including Mode 3.

Mode 4

The Bases for TS 3.4.10 state the pressurizer safety valves (PSV) are conservatively required to be operable in Mode 3 and portions of Mode 4 although the listed accidents may not require the PSVs for protection. TS 3.4.10 conservatively requires that all three PSVs be operable in Mode 4, even though Mode 4 operating conditions are significantly less limiting such that there are no design basis events which require explicit safety analysis to establish RCS overpressure protection requirements. The proposed note to TS 3.4.10 clarifies that for Mode 4 conditions, the RCS overpressure protection function is met as long as there is automatic pressure relief available as provided by the PSVs or a Class 1 PORV. Since there are no design basis events in Mode 4 which require a safety analysis and the postulation of a single failure, one

Class 1 PORV is adequate to establish RCS overpressure protection and minimize challenges to the PSVs.

TS 3.4.10 Note

The note added to TS 3.4.10 is intended to provide protection while not forcing the plant to be placed in a lower mode without both PORVs being operable or available. In Mode 3, one of the two Class I PORVs can be inoperable for 72 hours. The note allows for the PORV(s) to be determined operable to meet TS 3.4.11 prior to Mode 3 entry during plant startup, or to meet TS 3.4.12 prior to the RCS temperature decreasing below the LTOP arming temperature during a plant shutdown, where the PORVs are required to provide overpressure protection. Based on the limited time the plant would be in Mode 4 during startup or shutdown, the proposed change is justified.

NRC Question 2

The licensee indicated in its submittal that the safety grade PORVs are also credited in other design basis accidents. Please provide the following:

(a) A list of other design basis events using PORVs for accident mitigation. Explain why they are qualified for your evaluation in accordance with 10 CFR 50.59 and do not require the staff review.
(b) Address the staff concern of a single failure of the mitigating system during those design basis events assuming only one operable PORV required by the proposed TS 3.4.10.

PG&E Response

(a) In addition to the spurious safety Injection (SSI) event discussed in License Amendment Request (LAR) 01-08, the Diablo Canyon Power Plant (DCPP) license basis credits the Class 1 (safety-related) PORVs for mitigation in two other design basis events. These two events are the steam generator tube rupture (SGTR), and LTOP events, and both have already received NRC approval. The PORVs are also credited for mitigation in three additional evaluated events, which are included in this response for completeness. These events are associated with the steam generator alternate repair criteria, Appendix R fire protection safe shutdown criteria, and the anticipated transients without scram (ATWS) mitigating system actuation circuitry (AMSAC). The steam generator alternate repair criteria and the ATWS analysis have both received NRC approval, while the Appendix R fire protection safe shutdown credit for the PORV was incorporated into the DCPP licensing basis per 10 CFR 50.59.

The following summary provides a more detailed discussion of the design basis and beyond design basis accidents that credit the pressurizer PORV for mitigation and the associated documentation that incorporated these events into the DCPP

licensing basis.

Steam Generator Tube Rupture

PG&E credits the manual actuation of the PORVs to depressurize the RCS as one of the operator actions required to mitigate the SGTR accident. In PG&E Letter DCL-88-114, "Steam Generator Tube Rupture (SGTR) Analysis," dated April 29,1988, PG&E submitted the revised SGTR analysis using the NRC accepted Westinghouse Owners Group methodology established in WCAP-10698. The plant specific SGTR analysis for DCPP was documented in WCAP-11723 and provided as an enclosure to DCL-88-114. In Section C.4, WCAP-11723 states that the operator action to depressurize the RCS is accomplished using the pressurizer PORVS, when the reactor coolant pumps (RCP) are not running and pressurizer sprays are not available.

In NRC Letter dated April 3, 1991, "Closeout of Steam Generator Tube Rupture Analysis Issue for Diablo Canyon Power Plant, and Finding of Compliance with Condition 2.C.(9) of Unit 2 Operating License DPR-82," the enclosed safety evaluation (SE) confirmed that the DCPP PORVs primary motive power was non safety-related instrument air, but acknowledged the availability of safety grade back air supply. The SE confirmed the list of equipment credited for SGTR mitigation to be acceptable.

In PG&E Letter DCL-01-115, "License Amendment Request 01-05, Revision to Technical Specification 1.1, 'Definitions, Dose Equivalent I-131,' and Revised Steam Generator Tube Rupture and Main Steam Line Break Analyses," PG&E submitted LAR 01-05 to revise the dose consequences for the SGTR accident and identified in Section 4.1.2 that the pressurizer PORVs are credited to depressurize the RCS since the limiting case assumes the RCPs are tripped and pressurizer spray is not available. The NRC accepted this dose analysis in License Amendments (LA) 156 (Unit 1) and 156 (Unit 2) dated February 20, 2003.

Low Temperature Overpressure Protection

TS 3.4.12, "Low Temperature Overpressure Protection (LTOP) System," requires two of the three PORVs to be Class 1 rated for LTOP pressure protection. All of the PORVs are air operated, and the two safety-related PORVs have a nitrogen gas backup to the non safety-related air supply. This credited function of the PORVs was accepted by NRC as documented in LAs 133 (Unit 1) and 131 (Unit 2) and the associated SE dated May 3, 1999.

Steam Generator Alternate Repair Criteria

In PG&E Letter DCL-97-034, "License Amendment Request 97-03, Voltage-Based Alternate Steam Generator Tube Repair Limit for Outside Diameter Stress

Corrosion Cracking at Tube Support Plate Intersections," dated February 26, 1997, PG&E confirmed that DCPP was in compliance with the recommendations identified in Generic Letter (GL) 95-05, "Voltage-Based Repair Criteria For Westinghouse Steam Generator Tubes Affected By Outside Diameter Stress Corrosion Cracking." These recommendations are listed in Attachment 1 of GL 95-05 which is entitled; "Guidance for a Proposed License Amendment to Implement an Alternate Steam Generator Tube Repair Limit for Outside Diameter Stress Corrosion Cracking at the Tube Support Plate Intersections." Attachment 1, Section 2, "Tube Integrity Evaluation," discusses crediting the pressurizer PORVs for limiting the maximum primary to secondary pressure during a main steam line break (MSLB) event. GL 95-05, Attachment 1, Section 2 states:

> "For plants in which the TS do require the PORVs to be operable, the assumed differential pressure for the conditional burst probability calculation may be based on the PORV setpoint in lieu of the safety valve setpoint with similar adjustments. The TS requirements for operation with PORV block valves closed due to leaking PORVs should be in accordance with Enclosure A of Generic Letter 90-06, "Resolution of Generic Issue 70, 'Power-Operated Relief Valve and Block Valve Reliability,' and Generic Issue 94, 'Additional Low-Temperature Overpressure Protection for Light-Water Reactors,' Pursuant to 10 CFR 50.54(f)." That is, electrical power to the block valves must be maintained to allow continued operation with the block valves closed, as required in the sample technical specification Section 3.4.4 of GL 90-06."

This credit for PORV mitigation was incorporated into the DCPP design and licensing basis upon NRC acceptance documented in LAs 124 (Unit 1) and 122 (Unit 2) dated March 12, 1998.

Appendix R Safe Shutdown

In response to GL 81-12, "Fire Protection Rule" and GL 86-10 "Implementation of Fire Protection Requirements," PG&E established that the pressurizer PORVs would be available to perform a controlled RCS depressurization in order to achieve safe shutdown conditions as defined in 10 CFR 50 Appendix R. RCS overpressure protection for the Appendix R safe shutdown scenario is provided by the PSVs. Since NRC acceptance of these generic letter responses was not required, crediting the pressurizer PORVs for the Appendix R safe shutdown scenario was incorporated into the DCPP design basis per 10 CFR 50.59.

Anticipated Transients Without Scram Mitigating System Actuation Circuitry

Automatic actuation of the PORVs is credited in the generic ATWS analysis for AMSAC provided in topical report WCAP-10858-P-A, Revision 1, which was referenced in PG&E Letter DCL-87-258, "Plant-Specific AMSAC Design," dated

October 30, 1987. NRC acceptance of the DCPP AMSAC design was provided in NRC Letter dated August 15, 1988, "Safety Evaluation of the AMSAC System, PG&E's Proposed Method of Implementing the Requirements of 10 CFR 50.62 (ATWS) for Diablo Canyon."

(b) The proposed note to TS 3.4.10 that would require one Class 1 PORV to be available, is only applicable to Mode 3 and Mode 4 conditions. As discussed in the response to Question 1, for Mode 4 conditions the RCS overpressure protection function is met as long as there is automatic pressure relief available as provided by the PSVs or a Class 1 PORV. Since there are no design basis events in Mode 4 which require that a single failure safety analysis be considered, one Class 1 PORV is acceptable to establish RCS overpressure protection.

As discussed in section (a) of this response the only design basis events that could occur in Mode 3, and that credit the pressurizer PORVs for mitigation, are the SGTR and the SSI events. The note being added to TS 3.4.10 is considered acceptable for Mode 3 based on clarifying the technical difference between the applicable terms "operable" and "available for automatic actuation." Per TS 3.4.11 all three PORVs must be operable in Mode 3, however only one Class 1 PORV is required to be unblocked and available for automatic actuation which is consistent with the proposed Note to TS 3.4.10. The SGTR and the SSI events both credit operator action to make sure both Class 1 PORVs are unblocked and available for automatic actuation. The SGTR analysis which was approved in LAs 156/156, and the SSI analysis results and the operator simulator demonstration times submitted as part of LAR 01-08 have established that one PORV is capable of mitigating the SGTR and SSI events and that there is adequate time available for the operators to make both Class 1 pressurizer PORVs available. Therefore, in the event of the single failure of one PORV, or the associated block valve, the other Class I PORV is still available to mitigate the event.

NRC Question 3

The current TS allows PORVs to be isolated by their associated block valves to stop excessive seat leakage during power operation. Discuss the plant Emergency Operating Procedures (EOPs) that provide guidance to reactor operators for using those PORVs in accident mitigation using plant indications.

In a follow-up clarification received on October 1, 2003, the NRC staff asked that the EOP discussion include the guidance for making the PORVs available for both a SGTR event and an SSI event.

PG&E Response

As discussed in the response to Question 2, only two events during power operation, the SGTR and SSI events, credit operator action to unblock the PORVs and make them available for automatic actuation. The following applicable EOP procedural steps that the operators would use are summarized below:

The mitigation of the SGTR event is addressed through implementation of:

EOP E-0, "Reactor Trip or Safety Injection," Step 9.d requires checking power available to at least one block valve. Step 9.e requires checking at least one block valve open.

EOP E-3, "Steam Generator Tube Rupture," Step 12 requires operators to make power available to all three PORV block valves. Step 21.a requires making at least one PORV available. Step 21.b requires opening one PORV.

EOP ECA-3.3, "SGTR Without Pressurizer Pressure Control," provides additional guidance to operators for an event in which the operators are unable to open a PORV. Step 4 of this procedure provides operator guidance for the restoration of the pressurizer PORVs.

The mitigation of the SSI event is achieved through implementation of two procedures, EOP E-0 and EOP E-1.1 "Safety Injection Termination." EOP E-0 Step 9 requires that the operators unblock a PORV. This step was purposefully moved up within E-0 (rather than waiting for the transition to E-1.1), and the operators have been correspondingly trained in order to ensure that the two Class I PORVs are made available for automatic mitigation as soon as possible.

NRC Question 4

TS 3.4.11 requires that all PORVs shall be operable during Modes 1, 2 and 3. The proposed change to TS 3.4.10 seems to contradict this requirement during Mode 3 operation.

In a follow-up clarification received on October 1, 2003, the NRC staff stated they expect the response to address the basis for not requiring PORVs to be operable during Mode 4 with temperature higher than LTOP arming temperature. To use a reason such as "the events needing PORVs for accident mitigation are not likely to occur" is not appropriate since Modes are defined by RCS temperature and there is no assurance that the RCS pressures will be in sufficiently low in Mode 4 to assure a SGTR event will not occur. Also, a probability argument should not be applied in meeting a regulation.

PG&E Response

The apparent contradiction referenced in this question can be resolved based on the technical difference between the applicable terms "operable" and "available for automatic actuation." Per TS 3.4.11 all three PORVs must be operable, however only one needs to be unblocked and available for automatic actuation per the proposed change to TS 3.4.10. Having one PORV available is consistent with the single failure criteria for the SGTR and SSI analyses which are the only Mode 3 design basis events that credit the Class 1 PORV for mitigation. Therefore, the proposed text in TS 3.4.10 and TS 3.4.11 are not contradictory and are consistent with the assumptions in the SGTR and SSI analyses.

As discussed in the response to Question 2, the only design basis events that require the Class 1 PORVS to meet the single failure criteria are the SSI and the SGTR events. However, neither these design basis events nor any other design basis events which would require the actuation of the Class 1 PORVs for mitigation are considered credible in Mode 4 with the RCS cold leg temperature greater than LTOP arming temperature. DCPP Operating Procedures OP L-1, "Plant Heatup From Cold Shutdown to Hot Standby," and OP L-5, "Plant Cooldown From Minimum Load to Cold Shutdown," require that while in Mode 4, with the RCS cold leg temperature greater than LTOP arming temperature, the RCS must not exceed 350°F while the RCS pressure is maintained at 1000 psig. Under these conditions the automatic safety injection signal is still blocked such that it is not considered credible that a SSI event could occur. Also at these low RCS pressures, it is not considered credible that a design basis SGTR event could occur. However, any significant primary to secondary leakage which might occur during these Mode 4 conditions, would cause the RCS pressure to rapidly decrease and equilibrate with the steam generator pressure, since there would be no safety injection signal generated and no emergency core cooling system flow to maintain the RCS pressure at some intermediate elevated value. Such an event is not considered a significant challenge and the Class I PORVS would not be required to stabilize plant conditions.

NRC Question 5

Figure 3 of the licensee's submittal indicates that the pressurizer will become water solid in all three cases of the analyses. Please discuss the design adequacy of the PORVs, the associated tail piping and relief tank during this event.

PG&E Response

NRC Letter dated January 27, 1986, "Safety and Relief Valve Testing, NUREG-0737 Item II.D.1," issued the staff safety evaluation report (SER) for safety and relief valve testing for DCPP Units 1 and 2, in accordance with NUREG-0737, Item II.D.1. The SER provided acceptance of the adequacy of PORV and block valve design and confirmatory testing for a range of fluid conditions (full pressure steam, steam to water

transition, and subcooled water fluid) and acceptance of qualification to function properly during and following licensing basis seismic events. The SER also provided acceptance of the safety and relief valve piping between the pressurizer nozzles and the pressurizer relief tank (PRT). The adequacy of the PRT is discussed in Final Safety Analysis Report Update (FSARU) Section 5.5.10, "Pressurizer Relief Tank."

Note that LAR 01-08 proposes no changes to the design of the PORVs or related piping systems. The current approved mechanical design will remain as is. PG&E has upgraded the Instrument Class II portion of the PORV automatic actuation circuitry for Unit 2 and will complete the upgrade for Unit 1 at the next refueling outage. These upgrades were evaluated and determined not to require prior NRC approval in accordance with 10 CFR 50.59.

NRC Question 6

Verify the capacity certification for the plant PORVs in accordance with the requirements of Section III of the ASME Code.

PG&E Response

The pressurizer PORVs were procured and manufactured in accordance with USAS (now known as ANSI) B16.5. The requirements for these valves were established in a vendor equipment specification. There were no requirements for capacity certification in the specification. The design requirements for these valves are discussed in FSARU section 5.2 and the applicable codes for these valves are provided in Table 5.2-2.

The DCPP design as summarized in FSARU Table 5.5-16 contains three PORVs, two Class I, and one non-Class I, each with a rated flow capacity equal to 50 percent of a safety valve or 210,000 lb/hr at 2350 psig. NUREG-0737, Item II.D.1 required all pressurized-water reactor plant licensees and applicants to demonstrate that their pressurizer safety valves, PORVs, PORV block valves, and all associated discharge piping will function adequately under conditions predicted for design basis transients and accidents. This requirement was met as documented in the NRC SER issued on January 27, 1986 (see response to Question 5 above).

NRC Additional Question 6

In a follow-up response received on October 1, 2003, the NRC staff stated that the staff requested that the licensee verify the capacity certification for the plant PORVs in accordance with ASME Section III. The licensee's draft response is that the PORVs did not have any capacity certification required since they were not procured or manufactured to such requirements. The licensee also states that the staff's review of NUREG-0737, Item II.D.1 found that these valves would function adequately for design basis transients and accidents. However, the II.D.1 requirements were not intended as

a replacement or substitute for the ASME Section III requirements. The DCPP Unit 1 and Unit 2 reactor pressure vessels were designed to meet the requirements of the 1965 and 1968 Section III Codes (for Units 1 and 2 respectively), and are required to have overpressure protection in accordance with Section III. If the PSVs are incapable of providing overpressure protection, some other device which meets Section III must provide it. A review of the 1965 Code, for example, reveals that, in addition to other specific requirements, a device must: (a) have no stop valves unless they are positively controlled and interlocked, (b) not take advantage of external energy sources unless the device will properly function by self-actuation if the energy source fails, and (c) be manufactured, tested, certified, and marked, as required. The licensee should address all aspects of the ASME Section III requirements regarding credit for the PORV capacity, or alternatives should be justified in accordance with 10 CFR 50.55a(a)(3).

<u>**PG&E Response**</u>

The PSVs fulfill the RCS overpressure protection function as required by the ASME Code Section III, for all applicable Modes of operation. The DCPP design basis requirements for RCS overpressure protection are established in WCAP-7769 Rev. 1, "Topical Report Overpressure Protection for Westinghouse Pressurized Water Reactors," June 1972, which documents the compliance to the overpressure protection requirements of the ASME Boiler and Pressure Vessel Code Section III NB-7300 and NC-7300. The WCAP-7769 establishes the PSV relief capacity to be consistent with the ASME Section III requirement to maintain the RCS pressure less than 110 percent of the design value for the most limiting transient that occurs during normal operating conditions. The WCAP-7769 and the ASME Code Section III do not discuss or define Modes of operation as established in TS as a function of RCS temperature. However, since the limiting RCS overpressure analysis is a loss of load / loss of normal feedwater event from full power operating conditions, this establishes Mode 1 as the limiting TS mode of operation with respect to relief capacity. Similarly, the PSV loop seal hydraulic load design and associated loop seal temperature requirements, are also established based on the maximum relief discharge capacity for a Mode 1 limiting loss of load / loss of normal feedwater event. Since the DCPP TS require that all three PSVs (and their associated loop seals) be operable in Modes 1 and Mode 2, the PSV relief capacity remains bounded by the WCAP-7769 analysis and meets the ASME Code requirements for these Modes.

The automatic actuation of the Class I PORVs is not being credited and is not required to ensure RCS overpressure protection requirements are met in Modes 3 and 4 as specified in the ASME Code Section III. The automatic actuation of the Class 1 PORVs is only credited to prevent liquid relief through the PSVs during a Condition II SSI event in Mode 3, and to minimize any potential challenges to the PSV opening in Modes 3 and 4 with the loop seal temperature less than the TS required value. The concern with PSV opening during liquid relief conditions or when the loop seal temperature is less than the TS required value, is the ability to ensure the valve reseats properly and no leakage occurs after the valve closes. The WCAP-7769 and the ASME Code

Section III only establish the required relief valve capacity with respect to maximum full power operating conditions, and do not specifically require or address RCS overpressure analyses be performed for lower modes of operation. Since the required relief capacity and hydraulic loads for these lower modes and operating conditions would be much less than the maximum design values, the PSVs are still capable of opening to provide adequate RCS overpressure protection in Modes 3 and 4. In summary, a loop seal temperature less than the TS required value does not prevent a PSV from fulfilling the required ASME overpressure protection function, yet it is still conservatively prudent to minimize any such potential PSV opening in Modes 3 and 4 by maintaining a Class 1 PORV available for automatic actuation.

<u>NRC Question 7</u>

In accordance with Section III of the ASME Code, verify that only ½ of the total certified capacity of the available number of PORVs has been credited for applicable overpressure events in the plant safety analysis.

<u>PG&E Response</u>

PG&E credits one PORV for SSI mitigation, equal to one-third of the total PORV capacity, and equal to one-half of the Class I PORV capacity.

<u>NRC Additional Question 7</u>

In a follow-up response received on October 1, 2003, the NRC staff stated the staff requested that the licensee verify that only ½ of the total certified capacity of the PORVs is credited. The licensee's draft response is that the capacity of only one valve is credited. Section III requires that at least two valves always be available and restricts the use of stop valves, as discussed above. The licensee wishes to credit operator action to unblock enough valves to assure that two valves are available. However, this is not consistent with Section III. Also, it is not clear whether the capacity the licensee refers to is the certified capacity required by Section III.

<u>PG&E Response</u>

As discussed in the response to Question 6, the automatic actuation of the Class I PORVs is not being credited and is not required to ensure RCS overpressure protection requirements are met in Modes 3 and 4 as specified in the ASME Code Section III. Therefore, the Class 1 PORVs are not required to meet the ASME Section III requirements with respect to certified relief capacity, or the use of block valves.

NRC Question 8

Verify that the plant PORVs have been qualified to function properly during and following licensing basis seismic events.

PG&E Response

As stated in the response to Question 5, NRC Letter dated January 27, 1986, "Safety and Relief Valve Testing, NUREG-0737 Item II.D.1," issued the SER for safety and relief valve testing for DCPP Units 1 and 2, in accordance with NUREG-0737, Item II.D.1. The SER provided acceptance of the adequacy of PORV and block valve design and confirmatory testing for a range of fluid conditions (full pressure steam, steam to water transition, and subcooled water fluid) and acceptance of qualification to function properly during and following licensing basis seismic events. The SER also provided acceptance of the safety and relief valve piping between the pressurizer nozzles and the PRT.

NRC Additional Question 8

In a follow-up response received on October 1, 2003, the NRC staff stated that the staff requested that the licensee verify the functional seismic qualification of the PORVs. The licensee's draft response is that the staff's SER for Item II.D.1 provided acceptance of the adequacy of the PORVs and block valves to function during and following seismic events. Item II.D.1 required that PORVs and block valves be qualified by full flow testing to function for design basis transient and accident fluid inlet conditions. The test program also imposed large moments on the valve inlet and outlet which bounded the maximum moments in the plant configuration. In addition, the valve inlet and discharge piping (including the valve pressure boundary components) were analyzed for the plant maximum fluid induced loads, and the design basis seismic loads were included in the piping analysis. The extended masses of the valve bonnets and actuators were also modeled in the analysis. However, this analysis did not demonstrate that the PORV and block valve internal operating parts and actuators would function during and following seismic events. Since the PORVs and block valves are to be credited with the safety related overpressure relief function, the licensee should demonstrate the seismic functional qualification of these valves in accordance with GDC 2.

PG&E Response

The PG&E equipment seismic qualification program classifies as "active" valves, those that are required to perform their intended safety function during and/or after the most limiting seismic event. The DCPP PORVs and their block valves, like all active valves, have been qualified for seismic function via calculation(s) performed by the valve vendor and/or PG&E. The qualification calculations address the entire valve, including internals and appurtenances, as it will be installed in the piping system. Conformance

of the installed valves with the conditions and qualities reflected in these analyses is maintained by limiting the accelerations experienced by said valve(s) to values specified in the qualification calculation(s).

NRC Question 9

It is not clear how the crediting of PORV capacity for certain Mode 3 or 4 events reduces the loading conditions on the pressurizer safety valve (PSV) discharge piping. An analyzed event is the inadvertent opening of a PSV; therefore, the PSVs should be postulated to open spuriously, whether or not the PORV capacity is credited. Provide justification for not considering the loads from spuriously opening PSVs during the applicable Modes 3 or 4 conditions.

PG&E Response

FSARU Section 15.2.13 "Accidental Depressurization Of The Reactor Coolant System," states that the bounding accidental RCS depressurization event is based on the spurious opening of a PSV. This is an event which is analyzed at power to assure that the event does not result in the departure from nucleate boiling ratio (DNBR) decreasing below the minimum safety analysis limit. Since this event is limiting based on analyzing a maximum decrease in RCS pressure, the PSV safety function of RCS overpressure protection is not required for mitigation. The analysis is performed assuming bounding core kinetics and limiting Mode 1 reactor power conditions. This event is not specifically evaluated for conditions other than Mode 1.

The design basis of the pressurizer loop seal and discharge piping is to ensure that the PSV can successfully open and close as needed to mitigate a limiting RCS overpressure condition. The hydraulic loads on the discharge piping associated with a spurious opening of a PSV in Mode 3 or 4 conditions do not represent an adverse impact on any safety function required for mitigation such that the accidental RCS depressurization event currently analyzed in the FSARU remains bounding.

The adequacy of the design of the PSV and PORV discharge piping was included in the Westinghouse report "Pressurizer Safety and Relief Line Evaluation Summary Report - AM-SSA-2534, S. O. PGE/145" submitted by PG&E Letter from Philip A. Crane, Jr., to Harold R. Denton (NRC) dated December 13, 1982, and referenced in the NRC SER issued January 27, 1986, discussed above.

NRC Additional Question 9

In a follow-up response received on October 1, 2003, the staff noted that one of the stated reasons for crediting the PORVs for overpressure protection during Modes 3 and 4 is that the cooler loop seal loads are unacceptably high if a PSV discharges during these conditions. The staff requested that the licensee justify not considering the loads from spuriously opening PSVs during Modes 3 or 4, whether or not the PORVs are

credited for overpressure protection. The licensee's draft response is that the inadvertent PSV actuation event described in FSARU Section 15.2.13 only considers Mode 1 conditions because the event is only analyzed to evaluate DNBR. The licensee indicates that the overpressure protection function is not required during a decrease in RCS pressure. However, the licensee should address the greater challenge to the integrity of the PSVs and inlet and outlet piping which apparently occurs during an inadvertent discharge during Modes 3 or 4, not Mode 1.

PG&E Response

There is no explicit design requirement that the PSV discharge piping be able to withstand the structural loading associated with an inadvertent opening of a PSV in Modes 3 or 4. As discussed in the response to Question 9, the design basis of the discharge piping is to ensure that the PSV can effectively open and close to mitigate the limiting Condition II RCS overpressure event. An inadvertent or failed open PSV would cause a significant and unisolable RCS decrease, and the PSV discharge flow would be significantly greater than that required to cause the PRT rupture discs to open. This event would not require RCS overpressure protection, and since any potential loading on the discharge piping cannot make the event more severe, a structural evaluation for these conditions is not required.

NRC Question 10

It appears that a more direct and advantageous method of reducing the loads on the PSV discharge piping would be to eliminate the need for a loop seal upstream of the valves by refitting the PSVs with steam trim internals, which have a reduced tendency for seat leakage. Provide a discussion which addresses the need to maintain the loop seals, instead of eliminating them to significantly reduce discharge loads.

PG&E Response

PG&E has considered several options to address the low loop seal temperature issue, including elimination of the loop seal under the pressurizer safety valves, but does not consider this a viable alternative. PG&E's understanding is that a steam seat is much more sensitive to safety valve nozzle loads than a water seat. PG&E has measured the nozzle loads in the past and has implemented changes to reduce them to the point the valves have a good history of no seat leakage. Due to DCPP's seismic requirements, PG&E does not believe that the piping support arrangement can be modified to reduce nozzle loads to the values necessary to assure no valve leakage with a steam seat.

NRC Question 11

By letter dated January 27, 1982, the staff provided a safety evaluation of NUREG-0737, Item II.D.1 for Diablo Canyon 1 and 2. Therein, the staff found that according to EPRI report NP-2296 dated January 1981, the pressurizer would not become water solid until after 20 minutes following a spurious actuation of high pressure injection. The staff also found that this provided ample time for operator action to terminate the injection event and prevent water discharge through the PORVs and PSVs. The September 24, 2002, submittal states that 603^{0}F water could discharge through the PSVs in only 16 minutes. Provide a discussion to address this apparent inconsistency.

PG&E Response

As discussed in the background section of LAR 01-08, several additional issues have impacted the SSI analysis and resulted in reduced margin with respect to the time available for pressurizer overfill mitigation during a spurious safety injection event, as compared to the original analyses from EPRI Report NP-2296. Consequently, PG&E has had to (1) perform a much more detailed modeling of operator actions, and (2) upgrade the PORVs to Class I status in order to credit them for mitigation. A more detailed discussion of the historical evolution of the DCPP SSI analysis is provided below:

There have been several significant industry issues that have been identified since EPRI Report NP-2296 was published in January 1981. In Nuclear Safety Advisory Letter (NSAL) -93-013, "Inadvertent ECCS Analysis at Power," dated June 30, 1993, Westinghouse identified that the original analysis methodology might not be conservative for evaluating pressurizer overfill since it was established for verifying minimum DNBR limits. Soon after this, Westinghouse issued Supplement 1 to NSAL-93-013 dated October 28, 1994, which identified that the analysis methodology did not include modeling the operation of the nonsafety-related positive displacement pump which could lead to earlier pressurizer overfill times than previously analyzed. Based on Westinghouse indications that pressurizer overfill could occur as early as 10 minutes into the event, PG&E initiated a nonconformance report (N0001973) to address the issue.

The issues were finally resolved when Westinghouse issued a new SSI analysis (Westinghouse Letter PGE-96-565 dated May 31, 1996) that credited operator action to terminate the event within 16 minutes. The analysis also demonstrated that the minimum liquid temperature of 603.2°F, which was relieved through the PSVs for this event was acceptable. This analysis still represents the design basis analysis for pressurizer overfill that is referenced in FSARU Section 15.2.13. In December 1997, Westinghouse informed PG&E that a review of the FSARU analysis for a spurious operation of the safety injection system at power event identified that a temperature coefficient used for PSV modeling was treated incorrectly. The Westinghouse review determined the

temperature coefficient defined in WCAP-11677, "Pressurizer Safety Relief Valve Operation for Water Discharge During a Feedwater Line Break," January 1988, Appendix A is not a constant as previously treated, but rather varies with temperature. Previously, it was concluded that PSV operability would be maintained for a water relief temperature above 600°F. With the temperature coefficient correctly treated as a variable, the water temperature must remain above 613°F in order to justify stable PSV operation.

Westinghouse Letter PGE-98-502 dated January 15, 1998, identified that the pressurizer spray function could provide more effective pressure reduction and an earlier pressurizer overfill condition than previously analyzed. In order to resolve these outstanding issues, PG&E submitted DCL-99-071, "License Amendment Request (LAR) 99-01, Unreviewed Safety Question - Spurious Operation of the Safety Injection System at Power," dated May 21, 1999, which was based on revised Westinghouse analysis results that credited the pressurizer PORVs for mitigation of the SSI event and prevented opening of the pressurizer safety valves. After discussions with the NRC staff regarding the proposed LAR, PG&E withdrew the request pending review of other options.

In summary, numerous analysis and operational issues have been identified since the original EPRI report was published, which have significantly impacted the SSI analysis results. Therefore, for LAR 01-08, PG&E has performed a more detailed modeling of operator actions, and also upgraded the pressurizer PORVs to Class I in order to credit them for mitigation of the SSI event.

NRC Question 12

The following question was provided in a series of communications from June 5 to June 13, 2003, and is summarized as follows: How will actual operator action times for an SSI event be verified as conservatively bounded by the action times assumed in the SSI analysis?

PG&E Response

The operator action times assumed in the SSI analysis are time critical operator actions (TCOA) established and controlled by Administrative Procedure OP1.ID2, "Time Critical Operator Action." Operators will receive specific training (both initial and requalification) on the functional importance of performing these actions with respect to acceptable mitigation of the SSI event and ensuring that pressurizer overfill does not occur. Operating crew performance will be evaluated during an appropriate SSI scenario on the simulator and the recorded operator action times will be compared to the TCOAs assumed in the SSI analysis. If the recorded operator action times are not conservatively bounded by the TCOA times assumed in the analysis, then an evaluation must be performed to demonstrate acceptable results with respect to preventing pressurizer overfill condition, or the operator crews will be retrained on performing the appropriate actions within acceptable time frames.